**PSEG Nuclear LLC**
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
*Nuclear LLC*

March 26, 2007

10CFR50.73

LR-N07-066

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Hope Creek Generating Station Unit 1
Facility Operating License No. NPF-57
Docket No. 50-354

Subject: Licensee Event Report 2007-001-00

In accordance with 10 CFR 50.73(a)(2)(iv)(A), PSEG Nuclear LLC, is submitting Licensee Event Report Number 07-001-00, Docket No. 50-354.

Should you have any questions concerning this letter, please contact Mr. Francis D. Possessky at (856) 339-1160.

Sincerely,

John F. Perry

John F. Perry
Plant Manager
Hope Creek Generating Station

Attachment: Licensee Event Report

IE22

cc: Mr. S. Collins, Administrator - Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Ennis, Licensing Project Manager - Hope Creek
U.S. Nuclear Regulatory Commission
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Resident Inspector office - Hope Creek (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P.O. Box 415
Trenton, NJ 08625

| NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007 |
|---|---|---|

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)



| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
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**4. TITLE**
Low Reactor Water Level Scram

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | N/A | |
| 01 | 29 | 2007 | 2007 | - 001 - | 00 | 03 | 30 | 2007 | FACILITY NAME N/A | DOCKET NUMBER |

**9. OPERATING MODE:** 1

**10. POWER LEVEL:** 021

**11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§:** *(Check all that apply)*

| | | | |
|---|---|---|---|
| ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☒ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER |
| ☐ 20.2203(a)(2)(vi) | ☐ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Francis D. Possessky, Compliance Engineer | 856-339-1160 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| B | SJ | FE | F154 | Y | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* | ☒ NO | | | | |

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On January 29, 2007 during a plant startup from a planned outage with the reactor at 21% power and the main generator synchronized to the grid, an automatic reactor scram occurred due to low reactor water level.

The root cause of the scram was a failure of the instrument tap weld at the flow nozzle for the 'C' Reactor Feed Pump (RFP) Minimum Flow Line. This failure caused reduced indicated flow to the "C" RFP minimum flow control system logic and an increased minimum flow valve opening demand signal. As a result of these conditions reactor level could not be maintained and fell below the reactor low-level set point.

Corrective actions include repair of the affected weld and an extent of condition review to identify other potential weld deficiencies.

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17. TEXT (If more space is required, use additional copies of NRC Form 366A)

**PLANT AND SYSTEM IDENTIFICATION**

General Electric – Boiling Water Reactor (BWR/4)
Reactor Protection System – {JC}
Main Feedwater – {SJ}
Flow Element – {FE}

*Energy Industry Identification System {EIIS} codes and component function identifier codes appear as {SS/CCC}

**IDENTIFICATION OF OCCURRENCE**

Event Date/Time: January 29, 2007 - 23:12
Discovery Date/Time: January 29, 2007 - 23:12

**CONDITIONS PRIOR TO OCCURRENCE**

Hope Creek was in Operational Condition 1 with reactor power at approximately 21% during startup following a planned maintenance outage. No structures, systems, or components were inoperable that contributed to the event.

**DESCRIPTION OF OCCURRENCE**

Plant Startup was in progress with reactor power at 21% and reactor level in a stable oscillating pattern. Level ranged from 36 to 33.5 inches repeating every five minutes in a sinusoidal pattern. The "C" RFP was in service, manually maintaining a differential pressure of ~ 180 psid across the Start Up Level Control (SULC) valve. The SULC valve was in automatic, single element control maintaining reactor vessel level.
The "C" RFP total flow was operating in manual at between 4200 and 4800 gpm, with the minimum flow valve maintaining indicated pump total flow at greater than 5,000 gpm, the minimum flow automatic initiation set point. Indicated "C" RFP minimum flow ranged from zero to 800 gpm, as the system oscillated in response to injection demand.

Within three minutes of synchronizing the Main generator to the grid, reactor level began to lower out of the control band. The Licensed Plant Operator (PO) adjusted the 'C' RFP speed to control discharge pressure. Several speed adjustments were made by the PO in an attempt to maintain Reactor water level in the control band. During this level excursion, the PO returned the RFP speed signal to the previous setting in order to stabilize level. As Reactor Level continued to lower, the PO raised the speed signal to above the previous injection setting. The expected rise in discharge pressure was not observed. At this point the PO suspected an equipment failure associated with the 'C' RFP Minimum Flow valve, however, the Control Room indications showed the Minimum Flow at 0 gpm. Reactor Level continued to lower from 35" to 30" within 15 seconds and the Low Level Alarm was received at Level 4 (30"). The PO, having indications/response not as expected, attempted manual operation of the SULC valve. This had no effect due to a low differential pressure across the SULC valve, which provided little throttling. An attempt was made to place the 'A' RFP in service as Reactor Level continued to lower. Reactor Level lowered from 30" to 12.5" within one minute. As Reactor Water Level approached the Low Level setpoint, the Reactor Operator placed the Mode Switch in Shutdown, however the Reactor automatically scrammed on Low Water Level at 12.5" (Level 3) approximately 2 seconds prior to the insertion of the manual scram.

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17. TEXT (If more space is required, use additional copies of NRC Form 366A)

## CAUSE OF OCCURRENCE

The root cause of the scram was a bad weld at flow nozzle H1AE- 1AEFE-1770C. Lack of weld penetration to the tube wall allowed a leak path along the tube outside wall under the weld.

The leaking instrument tap weld provided input to instruments that supply "C" RFP minimum flow system flow indication to the CRIDS system, and indication and control feedback to the Foxboro Digital Feed Control System.

The failure reduced indicated flow to the "C" RFP minimum flow control system logic, which in turn increased valve opening demand, and resulted in significant excess flow through the "C" RFP minimum flow line.

Concurrent with this, as actual flow increased and significantly degraded pump performance, control room indication for minimum flow reduced to zero on both the Control Room Integrated Display System (CRIDS) and the Foxboro Digital Feed Control System (DFCS), and remained so throughout the bulk of the event.

Manual feed water control manipulation in response to the equipment failure was not effective in preventing the level from reaching the scram setpoint. Although manual action was taken to control RFPT speed, the SULC valve was left in the automatic level control mode. This configuration amplified the feed water system transient response, due to the SULC valve response not being in step with the operators' actions.

The effect of the line crack was also pressure dependent. At low pressures, the crack had little or no effect on system performance, but as pressure was raised above 650psig, the crack caused at first unstable indication, then as pressure approached normal operating pressure, indication became more and more erroneous.

## PREVIOUS OCCURRENCES

A review of previous reportable events at Hope Creek was performed to determine if a similar event had occurred. No similar events were identified.

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17. TEXT (If more space is required, use additional copies of NRC Form 366A)

## SAFETY CONSEQUENCES

The risk presented by this condition is minimal. The lowest recorded reactor level for the transient (-16") did not challenge ECCS set points.
A review of this event determined that a Safety System Functional Failure (SSFF) has not occurred as defined in Nuclear Energy Institute (NEI) 99-02.

## CORRECTIVE ACTIONS

The weld repair at flow nozzle H1AE-1AEFE-1770C was completed.

Penetrant Testing on each of the instrument tubing welds for flow indication associated with 'A', 'B' & 'C' RFP's will be performed during an outage.
Standardized methodology for responding to challenges from feedwater in low-power conditions will be developed.

## COMMITMENTS

This LER contains no commitments.